NATIONAL FOUNDATION ON THE ARTS AND THE HUMANITIES

National Endowment for the Arts

Federal Advisory Committee on International Exhibitions Advisory Meeting

Pursuant to Section 10(a)(2) of the Federal Advisory Committee Act (Public Law 92–463), as amended, notice is hereby given that a meeting of the Federal Advisory Committee on International Exhibitions will be held on September 28, 1995, from 9:00 a.m. to 5:30 p.m. This meeting will be held in Room M–07, at the Nancy Hanks Center, 1100 Pennsylvania Avenue, N.W., Washington, D.C. 20506.

Portions of this meeting will be open to the public from 9:00 a.m. to 9:15 a.m. for welcome and introductions and from 4:45 p.m. to 5:30 p.m. for a policy discussion.

The remaining portion of this meeting from 9:15 a.m. to 4:45 p.m. are for the purpose of Panel review, discussion, evaluation, and recommendation on applications for financial assistance under the National Foundation on the Arts and the Humanities Act of 1965, as amended, including information given in confidence to the agency by grant applicants. In accordance with the determination of the Chairman of June 22, 1995, this session will be closed to the public pursuant to subsection (c)(4), (6) and (9)(B) of section 552b of Title 5, United States Code.

Any person may observe meetings, or portions thereof, of advisory panels which are open to the public, and may be permitted to participate in the panel's discussions at the discretion of the panel chairman and with the approval of the full-time Federal employee in attendance.

If you need special accommodations due to a disability, please contact the Office of Special Constituencies, National Endowment for the Arts, 1100 Pennsylvania Avenue, N.W., Washington, D.C. 20506, 202/682–5532, TDY-TDD 202/682–5496, at least seven (7) days prior to the meeting.

Further information with reference to this meeting can be obtained from Ms. Yvonne Sabine, Committee Management Officer, National Endowment for the Arts, Washington, D.C. 20506, or call 202/682–5533.

Dated: September 7, 1995.

Yvonne M. Sabine,

Director, Office of Council and Panel Operations National Endowment for the Arts.

[FR Doc. 95–22648 Filed 9–12–95; 8:45 am]

BILLING CODE 7537–01–M

NUCLEAR REGULATORY COMMISSION

[Docket No. 50–302]

Florida Power Corporation; Notice of Denial of Amendment to Facility Operating License and Opportunity for Hearing

The U.S. Nuclear Regulatory Commission (the Commission) has denied a request by Florida Power Corporation, (licensee) for an amendment to Facility Operating License No. DPR–72 issued to the licensee for operation of the Crystal River Nuclear Generating Plant, Unit No. 3, located in Citrus County, Florida. Notice of Consideration of Issuance of this amendment was published in the **Federal Register** on November 14, 1990 (55 FR 47570).

The purpose of the licensee's amendment request was to revise the Technical Specifications (TS) to add a limiting condition for operation for new low temperature overpressure protection (LTOP) and to revise the reactor coolant system (RCS) heatup and cooldown pressure-temperature (PT) operating limits for operation up to 15 effective-full-power-years. On February 7, 1991, by Amendment No. 133, the NRC staff approved RCS heatup and cooldown PT curves for operation up to 15 effective-full-power-years. Amendment No. 133 did not address the licensee's proposed TS changes for LTOP, which is the subject of this notice.

The NRC staff has concluded that the licensee's request for LTOP TS changes cannot be granted. The licensee was notified of the Commission's denial of the proposed change by a letter dated August 31, 1995.

By October 13, 1995, the licensee may demand a hearing with respect to the denial described above. Any person whose interest may be affected by this proceeding may file a written petition for leave to intervene. A request for hearing or petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555, Attention: Docketing and Services Branch, or may be delivered to the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, by the above date.

A copy of any petitions should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555, and to A.H. Stephens, General Counsel, Florida Power Corporation, MAC–A5D, P.O. Box 14042, St. Petersburg, Florida 33733, attorney for the licensee.

For further details with respect to this action, see (1) the application for amendment dated October 31, 1989, as supplemented August 10, 1990, and (2) the Commission's letter to the licensee dated August 31, 1995.

These documents are available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, and at the local public document room located at the Coastal Region Library, 8619 W. Crystal Street, Crystal River, Florida 32629.

Dated at Rockville, Maryland, this 31st day of August 1995.

For the Nuclear Regulatory Commission.

David B. Matthews,

Project Director, Project Directorate II–1, Division of Reactor Projects—I/II, Office of Nuclear Reactor Regulation.

[FR Doc. 95–22703 Filed 9–12–95; 8:45 am]

BILLING CODE 7590–01–P

Biweekly Notice

Applications and Amendments to Facility Operating LicensesInvolving No Significant Hazards Considerations

I. Background

Pursuant to Public Law 97-415, the U.S. Nuclear Regulatory Commission (the Commission or NRC staff) is publishing this regular biweekly notice. Public Law 97-415 revised section 189 of the Atomic Energy Act of 1954, as amended (the Act), to require the Commission to publish notice of any amendments issued, or proposed to be issued, under a new provision of section 189 of the Act. This provision grants the Commission the authority to issue and make immediately effective any amendment to an operating license upon a determination by the Commission that such amendment involves no significant hazards consideration, notwithstanding the pendency before the Commission of a request for a hearing from any person.

This biweekly notice includes all notices of amendments issued, or proposed to be issued from August 18, 1995, through August 30, 1995. The last biweekly notice was published on Wednesday, August 30, 1995 (60 FR 45172).

Notice Of Consideration Of Issuance Of Amendments To Facility Operating Licenses, Proposed No Significant Hazards Consideration Determination, And Opportunity For A Hearing

The Commission has made a proposed determination that the following amendment requests involve no significant hazards consideration. Under the Commission's regulations in 10 CFR 50.92, this means that operation of the facility in accordance with the proposed amendment would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety. The basis for this proposed determination for each amendment request is shown below.

The Commission is seeking public comments on this proposed determination. Any comments received within 30 days after the date of publication of this notice will be considered in making any final determination.

Normally, the Commission will not issue the amendment until the expiration of the 30-day notice period. However, should circumstances change during the notice period such that failure to act in a timely way would result, for example, in derating or shutdown of the facility, the Commission may issue the license amendment before the expiration of the 30-day notice period, provided that its final determination is that the amendment involves no significant hazards consideration. The final determination will consider all public and State comments received before action is taken. Should the Commission take this action, it will publish in the **Federal Register** a notice of issuance and provide for opportunity for a hearing after issuance. The Commission expects that the need to take this action will occur very infrequently.

Written comments may be submitted by mail to the Rules Review and Directives Branch, Division of Freedom of Information and Publications Services, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555, and should cite the publication date and page number of this **Federal Register** notice. Written comments may also be delivered to Room 6D22, Two White Flint North, 11545 Rockville Pike, Rockville, Maryland from 7:30 a.m. to 4:15 p.m. Federal workdays. Copies of written comments received may be examined at the NRC Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC. The filing of requests for a hearing and petitions for leave to intervene is discussed below.

By October 13, 1995, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC and at the local public document room for the particular facility involved. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

As required by 10 CFR 2.714, a petition for leave to intervene shall set forth with particularity the interest of the petitioner in the proceeding, and how that interest may be affected by the results of the proceeding. The petition should specifically explain the reasons why intervention should be permitted with particular reference to the following factors: (1) the nature of the petitioner's right under the Act to be made a party to the proceeding; (2) the nature and extent of the petitioner's property, financial, or other interest in the proceeding; and (3) the possible effect of any order which may be entered in the proceeding on the petitioner's interest. The petition should also identify the specific aspect(s) of the subject matter of the proceeding as to which petitioner wishes to intervene. Any person who has filed a petition for leave to intervene or who has been admitted as a party may amend the petition without requesting leave of the Board up to 15 days prior to the first prehearing conference scheduled in the proceeding, but such an amended petition must satisfy the specificity requirements described above.

Not later than 15 days prior to the first prehearing conference scheduled in the proceeding, a petitioner shall file a supplement to the petition to intervene which must include a list of the contentions which are sought to be litigated in the matter. Each contention must consist of a specific statement of the issue of law or fact to be raised or controverted. In addition, the petitioner shall provide a brief explanation of the bases of the contention and a concise statement of the alleged facts or expert opinion which support the contention and on which the petitioner intends to rely in proving the contention at the hearing. The petitioner must also provide references to those specific sources and documents of which the petitioner is aware and on which the petitioner intends to rely to establish those facts or expert opinion. Petitioner must provide sufficient information to show that a genuine dispute exists with the applicant on a material issue of law or fact. Contentions shall be limited to matters within the scope of the amendment under consideration. The contention must be one which, if proven, would entitle the petitioner to relief. A petitioner who fails to file such a supplement which satisfies these requirements with respect to at least one contention will not be permitted to participate as a party.

Those permitted to intervene become parties to the proceeding, subject to any limitations in the order granting leave to intervene, and have the opportunity to participate fully in the conduct of the hearing, including the opportunity to present evidence and cross-examine witnesses.

If a hearing is requested, the Commission will make a final determination on the issue of no significant hazards consideration. The final determination will serve to decide when the hearing is held.

If the final determination is that the amendment request involves no significant hazards consideration, the Commission may issue the amendment and make it immediately effective, notwithstanding the request for a hearing. Any hearing held would take place after issuance of the amendment.

If the final determination is that the amendment request involves a significant hazards consideration, any hearing held would take place before the issuance of any amendment.

A request for a hearing or a petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555, Attention: Docketing and Services Branch, or may be delivered to the Commission's Public

Document Room, the Gelman Building, 2120 L Street, NW., Washington DC, by the above date. Where petitions are filed during the last 10 days of the notice period, it is requested that the petitioner promptly so inform the Commission by a toll-free telephone call to Western Union at 1-(800) 248-5100 (in Missouri 1-(800) 342-6700). The Western Union operator should be given Datagram Identification Number N1023 and the following message addressed to **(Project Director)**: petitioner's name and telephone number, date petition was mailed, plant name, and publication date and page number of this **Federal Register** notice. A copy of the petition should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555, and to the attorney for the licensee.

Nontimely filings of petitions for leave to intervene, amended petitions, supplemental petitions and/or requests for a hearing will not be entertained absent a determination by the Commission, the presiding officer or the Atomic Safety and Licensing Board that the petition and/or request should be granted based upon a balancing of factors specified in 10 CFR 2.714(a)(1)(i)-(v) and 2.714(d).

For further details with respect to this action, see the application for amendment which is available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, and at the local public document room for the particular facility involved.

Carolina Power & Light Company, Docket No. 50-261, H. B. Robinson Steam Electric Plant, Unit No. 2, Darlington County, South Carolina

Date of amendment request: July 17, 1995

Description of amendment request: The requested change to Technical Specification (TS) section 3.8 would specify that the spent fuel building refueling filter fan and at least one containment purge fan shall be shown to operate within plus or minus 10 percent of the design flow.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:The proposed change to TS is to revise Section 3.8.2.c. This TS section currently states ''All filter system fans shall be shown to operate within [plus or minus] 10% of the design flow.'' The proposed requirements are as follows:

c.1 The Spent Fuel Building refueling filter fan shall be shown to operate within [plus or minus] 10% of the design flow.

c.2 At least one Containment purge filter fan shall be shown to operate within [plus or minus] 10% of the design flow and must be operable during core alterations or movement of irradiated fuel assemblies, or at least one automatic containment isolation valve in each line penetrating the containment which provides a direct path from the containment atmosphere to the outside atmosphere shall be securely closed.

This proposed change does not involve a significant hazards consideration for the following reasons.

1. The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed change clarifies the operating requirements for the Containment purge and Spent Fuel Building refueling filter systems. This proposed change to the TS specifically delineates the fan filter systems required for refueling operations and does not change the physical operation of the filter systems. The affected systems are not involved in the initiation of any accident. The system response to previously analyzed accidents, including system flows and filter efficiencies will not be altered by the proposed change. These changes are enhancements to clarify existing TS requirements that will not increase the probability or consequences of a previously analyzed accident.

2. The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change merely clarifies the specific filter systems that are necessary to mitigate a fuel handling accident during core alterations or the movement of irradiated fuel assemblies and is consistent with the accident analysis in Section 15.7.4 of the Updated Final Safety Analysis Report (UFSAR). This proposed change does not involve the addition or modification of plant equipment, nor does it alter the design or operation of plant systems. Therefore, operation of the facility in accordance with the proposed TS change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed change clarifies which filter systems that must be capable of mitigating a design basis fuel handling accident during core alterations or the movement of irradiated fuel assemblies and is consistent with the accident analysis in Section 15.7.4 of the UFSAR. The proposed change will not result in an increase in the Control Room or offsite radiation doses. The performance of the filtration systems, including adsorption efficiencies, will not change. Therefore, the proposed change does not involve a reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Hartsville Memorial Library, 147 West College Avenue, Hartsville, SC 29550

Attorney for licensee: R. E. Jones, General Counsel, Carolina Power & Light Company, Post Office Box 1551, Raleigh, NC 27602

NRC Project Director: David B. Matthews

Commonwealth Edison Company, Docket Nos. 50-295 and 50-304, Zion Nuclear Power Station, Units 1 and 2, Lake County, Illinois

Date of amendment request: June 30, 1995

Description of amendment request: The proposed amendments would modify the emergency diesel generator testing requirements in the Technical Specifications.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. The proposed changes do not involve a significant increase in the probability or consequences of occurrence of any accident previously evaluated.

The proposed changes to the Technical Specifications will change the scope of EDG [Emergency Diesel Generator] testing that is performed on a refueling cycle frequency. The proposed changes will eliminate the requirement to perform sequenced loading of the EDG as part of the hot restart test, and will allow the hot restart test to be initiated from any EDG start signal. The revised requirements will eliminate testing that is redundant, provides no additional meaningful information, significantly constrains scheduling of refueling outage maintenance and testing, and impacts the availability of systems and components important to safety. The proposed testing requirements satisfy the underlying purpose of the EDG hot restart test. The testing in accordance with the proposed requirements will verify the ability of each EDG to complete the start up sequence from an equilibrium temperature immediately following operation at full load for a period of time long enough to stabilize operating temperature.

A two hour period for operation at full load has been chosen to ensure that full load operating temperature has stabilized prior to shutdown preceding the hot restart test. Momentary transients outside the full load operating band of 3600 to 4000 kW will not invalidate the two hour run since momentary transient will not significantly affect operating temperature. Brief operation subsequent to a momentary transient will normalize operating temperature. Since the proposed changes impact only surveillance requirements used to periodically verify the operability of a required safety system, and since the proposed changes provide an equivalent level of testing and eliminate redundant testing, the proposed changes will not impact the operability or availability of a required system.

Operation in accordance with the revised requirements will not increase the likelihood that a transient initiating event will occur since transients are initiated by equipment malfunction and/or catastrophic system failure. The revised requirements affect testing that is performed on a Refueling Cycle frequency. Testing in accordance with the proposed requirements will not increase the probability of failure of the EDGs since the testing will provide an equivalent level of testing to verify the operability of the EDGs. In addition, failure of an EDG to start or failure of an EDG while operating is not assumed to be an initiating event of an accident considered in the Updated Final Safety Analysis Report (UFSAR). Based on the above, operation in accordance with the proposed requirements will not significantly increase the probability of occurrence of any accident previously evaluated.

The proposed requirements will meet the underlying purposed of the existing testing requirements. The proposed testing will ensure the ability of the EDG to start from a hot condition in the unlikely event of an accident. The proposed changes will eliminate testing requirements that are redundant and unnecessarily challenge the reliability of the EDGs by requiring unnecessary wear and cycling of the diesel engine and auxiliary systems. Since the proposed changes will not adversely affect the operability or availability of the EDGs, the ability of the EDGs to operate and power equipment important to safety will not be impacted and the ability to mitigate the consequences of accidents previously evaluated will not be affected. Based on the preceding discussion, the consequences of accidents previously evaluated will not significantly increase.

2. The proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed changes to the Technical Specifications do not involve the addition of any new or different types of safety related equipment, nor do they involve the operation of equipment required for safe operation of the facility in a manner different from those addressed in the UFSAR. No safety related equipment or function will be altered as a result of the proposed changes. Also, the procedures that govern normal operation and recovery from an accident are not affected by the proposed changes.

The proposed changes will eliminate testing requirements that are redundant and provide no additional meaningful information. Testing in accordance with the revised requirements will provide an equivalent level of confidence in the reliability of the EDG systems to complete the start up sequence from a hot condition. The proposed testing requirements satisfy the purpose Regulatory Guide 1.108 in that the testing requirements will ensure EDG operability and reliability. In addition, the proposed changes are consistent with the changes recommended by the NRC in Generic Letter 93-05. Since no new failure modes or mechanisms are introduced by the proposed changes, the possibility of a new or different kind of accident is not created.

3. The proposed changes do not involve a significant reduction in a margin of safety.

Plant safety margins are established through LCOs, limiting safety system settings, and safety limits specified in the Technical Specifications. There will be no changes to either the physical design of the plant or to any of these settings or limits as a result of the proposed changes. The proposed changes will eliminate testing requirements that are redundant and provide no additional information. Testing in accordance with the revised requirements will verify the ability of the EDGs to complete the start up sequence from a hot condition as is intended by the recommended testing in Regulatory Guide 1.108. In addition, the proposed changes are consistent with the changes recommended by the NRC in Generic Letter 93-05. Since the proposed changes will not impact the availability or operability of the EDGs to perform their intended function and since no LCOs, safety limits, or safety system settings are affected by the proposed changes, there is no significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the requested amendments involve no significant hazards consideration.

Local Public Document Room location: Waukegan Public Library, 128 N. County Street, Waukegan, IL 60085

Attorney for licensee: Michael I. Miller, Esquire; Sidley and Austin, One First National Plaza, Chicago, IL 60603

NRC Project Director: Robert A. Capra

Florida Power and Light Company, Docket Nos. 50-250 and 50-251, Turkey Point Plant Units 3 and 4, Dade County, Florida

Date of amendment request: July 26, 1995.

Description of amendment request: The licensee proposes to change Turkey Point Units 3 and 4 Technical Specifications to allow rod misalignment of +/- 18 steps at or below 90% of rated thermal power. In addition, a change is proposed to the Limiting Condition for Operation range of rod travel from 228 to ''All Rods Out.'' The introduction of ''All Rods Out'' is consistent with Amendment 167/161 which approved the removal of Technical Specification 3.1.3.6, ''Rod Insertion Limit'' from the Technical Specifications and placement into the Core Operating Limits Report (COLR).

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

(1) Operation of the facility in accordance with the proposed amendment would not involve a significant increase in the probability or consequences of any accident previously evaluated.

The proposed limits on rod misalignment do not increase the probability of an accident. The Technical Specifications' allowed increase in peaking factor limits as power is reduced accommodates an increase in rod misalignment of [plus or minus] 18 steps below 90% of RTP [rated thermal power]. The initial conditions remain unchanged from that assumed in the Updated Final Safety Analysis Report (UFSAR). Therefore, this proposed change poses no significant increase in the probability or consequences of any accident previously evaluated.

(2) Operation of the facility in accordance with the proposed amendment would not create the possibility of a new or different kind of accident from any accident previously evaluated.

No new accident scenarios, failure mechanisms or limiting single failure are introduced as a result of implementing the proposed rod misalignment criteria. The institution of the proposed rod misalignment criteria will have no adverse effect, nor does it challenge, the performance of any other safety related system. Therefore, the proposed amendment does not in any way create the possibility of a new or different kind of accident from any accident previously evaluated.

(3) Operation of the facility in accordance with the proposed amendment would not involve a significant reduction in the margin of safety.

Operation of the facility in accordance with the proposed amendment would not involve a significant reduction in the margin of safety. The margin of safety, as defined in the BASES for the Technical Specifications, is not significantly affected by the changes to the rod misalignment limit. The Technical Specifications' allowed increase in peaking factor limits as power is reduced accommodates an increase in rod misalignment of [plus or minus] 18 steps below 90% of RTP. The initial conditions remain unchanged from that assumed in the UFSAR. Since the peaking factor limits are not modified, the proposed change does not constitute a significant reduction in the margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Florida International University, University Park, Miami, FL 33199

Attorney for licensee: J. R. Newman, Esquire, Morgan, Lewis & Bockius, 1800 M Street, NW., Washington, DC 20036

NRC Project Director: David B. Matthews

Florida Power and Light Company, Docket Nos. 50-250 and 50-251, Turkey Point Plant Units 3 and 4, Dade County, Florida

Date of amendment request: July 26, 1995.

Description of amendment request: The licensee proposes to change Turkey Point Units 3 and 4 Technical Specifications to delete the requirement to adjust the Nuclear Instrumentation System (NIS) downward when operating at less than 70% of rated thermal power (RTP).

At reduced power levels (i.e., less than 70% of RTP), calorimetric power measurement uncertainties are most influenced by the feedwater flow measurements, which have the potential for large flow uncertainties under low flow conditions. These calorimetric uncertainties create the potential for a non-conservative gain adjustment of the NIS when the NIS is adjusted downward to match calorimetric power at reduced power levels, and may result in a non-conservative NIS power level indication when operating at higher power levels. Inappropriate gain adjustments could cause the NIS Power Range High Neutron Flux trip to occur at power levels beyond that assumed in the plant safety analyses. The proposed changes would correct this situation.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

(1) Operation of the facility in accordance with the proposed amendment would not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed change does not involve any physical changes to the NIS. Implementation of the proposed change does not affect the probability of failure of the NIS and does not alter the method in which protection is afforded by the NIS for the reactor and primary system. Therefore, the proposed change does not result in an increase in the severity or consequences of any accident previously evaluated.

The proposed change in Technical Specifications to remove the requirement which could result in non-conservative gain adjustments of the NIS at reduced power levels (below 70% of RTP), will have no significant effect on the probability or consequences of licensing basis events; and the probability or consequences of an accident previously evaluated for Turkey Point has not been significantly increased. Therefore, operation of the facility in accordance with the proposed amendments would not involve a significant increase in the probability or consequences of an accident previously evaluated.

(2) Operation of the facility in accordance with the proposed amendment would not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed change does not result in a change in the method in which the NIS provides plant protection. No change is being made which alters the function of the NIS. Therefore, the proposed change does not create the possibility of a new or different kind of accident nor involve a reduction in a margin of safety as defined in the Safety Analysis Report.

The change in Technical Specifications associated with the removal of the requirement which could result in non-conservative gain adjustments of the NIS at reduced power levels (below 70% of RTP) will not create the possibility of a new or different kind of accident from any accident previously evaluated.

(3) Operation of the facility in accordance with the proposed amendment would not involve a significant reduction in the margin of safety.

This change in Technical Specifications only affects the removal of the requirement which has the potential for non-conservative gain adjustments of the NIS at reduced power levels (below 70% of RTP); these changes do not alter the manner in which protection is afforded for the reactor and primary system. In addition, the fundamental process for implementation of the calorimetric power/NIS comparison remains the same.

The changes in Technical Specifications associated with the removal of the requirement, which could lead to non-conservative gain adjustments of the NIS at reduced power levels (below 70% of RTP), will not involve a significant reduction in the margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Florida International University, University Park, Miami, FL 33199

Attorney for licensee: J. R. Newman, Esquire, Morgan, Lewis & Bockius, 1800 M Street, NW., Washington, DC 20036

NRC Project Director: David B. Matthews

Florida Power and Light Company, Docket Nos. 50-250 and 50-251, Turkey Point Plant Units 3 and 4, Dade County, Florida

Date of amendment request: July 26, 1995.

Description of amendment request: The licensee proposes to change Turkey Point Units 3 and 4 Technical Specifications (TS) to incorporate certain changes which are consistent with guidance provided by NUREG-1366 and NRC Generic Letter (GL) 93-05, ''Line-Item Technical Specification Improvements to Reduce Surveillance Requirements for Testing During Power Operation.'' The following proposed changes are requested:

(1) TS SR 4.1.3.1.2: Change the frequency interval for control rod movement test from monthly to quarterly.

(2) TS SR 4.6.5.1: Change the hydrogen monitor calibration from quarterly to each refueling interval, and the analog channel operational test from monthly to quarterly.

(3) TS SR Table 4.3-3: Change the analog channel functional test from monthly to quarterly for radiation monitors. Correct spelling of 'Radioactivity' in Item 1.a.

(4) TS SR 4.4.6.2.2: Increase the time allowed in COLD SHUTDOWN before leak testing the Reactor Coolant System (RCS) isolation valves is required, from 72 hours to 7 days.

(5) TS SR 4.10.1.2: Changes the requirement for a rod drop test prior to reducing SHUTDOWN MARGIN from ''within 24 hours'' to ''within 7 days''.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

(1) Operation of the facility in accordance with the proposed amendments would not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed amendments do not involve a significant increase in the probability or consequences of an accident previously evaluated because the proposed amendments conform to the guidance given in Enclosure 1 of the NRC Generic Letter 93-05. The overall functional capabilities of the rod control system, RCS pressure isolation valves, the hydrogen monitoring system, and the radiation monitoring systems will not be modified by the proposed change. These amendments will not involve a significant increase in the probability or consequences of an accident previously evaluated for the following reasons:

(1) Increasing the interval of control rod movement testing will reduce the possibility of testing-related reactor trips and dropped rods, and result in fewer challenges to safety systems and plant transients.

(2) Increasing the interval of hydrogen monitor calibration and operational tests will result in a reduction in equipment degradation and reduce a burdensome task on personnel resources.

(3) Increasing the interval of radiation monitor functional tests will result in less equipment degradation as well as reducing the potential for testing-related isolations of the control room, fuel handling building, auxiliary buildings, and various process lines.

(4) Increasing the time allowed in COLD SHUTDOWN prior to leak testing RCS isolation valves will permit plant personnel to focus on short notice outage recovery and minimize personnel radiation exposure. Since the methods and the acceptance criteria used for the leak test are not altered, increasing the time from 72 hours to 7 days will not significantly alter the associated risk.

(5) Increasing the time required to perform rod tests prior to reducing the SHUTDOWN MARGIN will result in only one rod drop test vice two following a refueling outage, which will in turn reduce plant transients and personnel resource requirements.

(2) Operation of the facility in accordance with the proposed amendments would not create the possibility of a new or different kind of accident from any accident previously evaluated.

The use of the proposed changes to the TS can not create the possibility of a new or different kind of accident from any accident previously evaluated since the proposed amendments will not change the physical plant or the modes of plant operation defined in the facility operating license. No new failure mode is introduced due to the surveillance interval changes and clarifications, since the proposed changes do not involve the addition or modification of equipment nor do they alter the design or operation of affected plant systems.

(3) Operation of the facility in accordance with the proposed amendments would not involve a significant reduction in a margin of safety.

The operating limits and functional capabilities of the affected systems are unchanged by the proposed amendments. The proposed changes to the TS which establish new or clarify old surveillance intervals consistent with the NRC Generic Letter 93-05 line-item improvement guidance do not significantly reduce any of the margins of safety even though the number of surveillances is decreased. These requested amendments are justified by the following reasoning from NUREG-1366:

(1) The surveillances could lead to plant transients which would challenge safety systems unnecessarily as in the cases of control rod movement tests and post-refueling rod drop tests.

(2) The surveillances result in the unnecessary wear to equipment as in the cases of the hydrogen and radiation monitor surveillances.

(3) The surveillance result in radiation exposure to plant personnel which is not justified by the safety significance of the surveillances as in the case of the time requirement for leak-testing RCS isolation valves when in COLD SHUTDOWN.

(4) The surveillances place an unnecessary burden on plant personnel because the time required is not justified by the safety significance of the surveillance, i.e. hydrogen monitor and post-refueling rod drop tests.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Florida International University, University Park, Miami, FL 33199

Attorney for licensee: J. R. Newman, Esquire, Morgan, Lewis & Bockius, 1800 M Street, NW., Washington, DC 20036

NRC Project Director: David B. Matthews

Florida Power and Light Company, Docket Nos. 50-250 and 50-251, Turkey Point Plant Units 3 and 4, Dade County, Florida

Date of amendment request: July 26, 1995.

Description of amendment request: The licensee proposes to change Turkey Point Units 3 and 4 Technical Specification Administrative Controls Section 6.9.1.7 to reflect the use of the Westinghouse NOTRUMP model in the Small Break Loss-of-Coolant Accident (SBLOCA) analysis used in determining the K(z) curve contained in the Core Operating Limits Report (COLR). The following references would be added to Section 6.9.1.7 (COLR) of the Administrative Controls section of Turkey Point Units 3 and 4 TS: ≥WCAP-10054-P-A, (proprietary) and WCAP-10081-NP-A, (non-proprietary), ''Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code'', October, 1985.'' WCAP-10054-P-A Addendum 2, (proprietary), ''Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection into the Broken Loop and COSI Condensation Model'', August, 1994.''

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

(1) Operation of the facility in accordance with the proposed amendment would not involve a significant increase in the probability or consequences of an accident previously evaluated.

The modification to the current Section 6.9.1.7 of the Administrative Controls section of the Turkey Point Technical Specifications to include the references to WCAP-10054-P-A, ''Small Break ECCS Evaluation Model Using the NOTRUMP Code'', and WCAP-10054-P-A Addendum 2 for the COSI model, does not involve an increase in the probability or consequences of an accident previously evaluated. This modification to the Technical Specification does not change the probability of occurrence previously evaluated.

This change does not affect the integrity of the fission product barriers utilized for mitigation of radiological dose consequences as a result of an accident. The addition of the new methodology used for Turkey Point uprating analysis does not change, degrade, or prevent the response of safety related mitigation systems to accident scenarios, as described in the Updated Final Safety Analysis Report (UFSAR) Chapter 14. Therefore, the licensee concluded that the probability or consequences of an accident previously evaluated are not increased.

(2) Operation of the facility in accordance with the proposed amendment would not create the possibility of a new or different kind of accident from any accident previously evaluated.

The modification to the current Section 6.9.1.7 of the Administrative Controls section of the Turkey Point Technical Specifications to include the references to WCAP-10054-P-A, ''Small Break ECCS Evaluation Model Using the NOTRUMP Code'', and WCAP-10054-P-A Addendum 2 for the COSI model, will not create the possibility of a new or different kind of accident from any accident previously evaluated. No new operating configuration is being imposed by the addition of the references to the Technical Specification. Therefore, no new failure modes or limiting single failures have been identified. The licensee concludes that no new or different kind of accidents from those previously evaluated have been created as a result of this revision.

(3) Operation of the facility in accordance with the proposed amendment would not involve a significant reduction in the margin of safety.

The modification to the current Section 6.9.1.7 of the Administrative Controls section of the Turkey Point Technical Specifications to include the references for the Small Break ECCS Evaluation Model Using the NOTRUMP Code will not involve a reduction in the margin of safety. The SBLOCA analysis results show that the limits of 10 CFR 50.46 are maintained as follows. The new calculated value of worst-case PCT will be 1688°F, which is less than the limit of 2200°F. There is significant margin in the current SBLOCA analysis such that the total cladding oxidation limit of 17 percent will not be challenged. Further, the calculated total amount of hydrogen generated has been determined to remain less than 1 percent. The SBLOCA hydraulic forces are not affected by the K(z) curve and the licensee concludes that the core will remain amenable to cooling. Additionally, post-LOCA long term core cooling and hot leg switchover evaluations are not impacted by the K(z) curve. Therefore, there is no significant reduction in the margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Florida International University, University Park, Miami, FL 33199

Attorney for licensee: J. R. Newman, Esquire, Morgan, Lewis & Bockius, 1800 M Street, NW., Washington, DC 20036

NRC Project Director: David B. Matthews

Florida Power and Light Company, Docket Nos. 50-250 and 50-251, Turkey Point Plant Units 3 and 4, Dade County, Florida

Date of amendment request: July 26, 1995.

Description of amendment request: The licensee proposes to change Turkey Point Units 3 and 4 Technical Specifications to achieve consistency throughout these documents by (a) removing outdated material, (b) incorporating administrative clarifications and corrections, and (c) correcting typographical errors.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

(1) Operation of the facility in accordance with the proposed amendment would not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed amendments do not involve a significant increase in the probability or consequences of an accident previously evaluated because the proposed amendments are purely administrative in nature. These amendments will not involve a significant increase in the probability or consequences of an accident previously evaluated because they do not affect assumptions contained in plant safety analyses, the physical design and/or operation of the plant, nor do they affect Technical Specifications that preserve safety analysis assumptions. Therefore, the proposed changes do not affect the probability or consequences of accidents previously analyzed.

(2) Operation of the facility in accordance with the proposed amendment would not create the possibility of a new or different kind of accident from any accident previously evaluated.

The use of the modified specifications can not create the possibility of a new or different kind of accident from any previously evaluated since the proposed amendments will not change the physical plant or the modes of plant operation defined in the facility operating license. No new failure mode is introduced due to the administrative changes and clarifications, since the proposed changes do not involve the addition or modification of equipment nor do they alter the design or operation of affected plant systems, structures, or components.

(3) Operation of the facility in accordance with the proposed amendment would not involve a significant reduction in the margin of safety.

The operating limits and functional capabilities of the affected systems, structures, and components are unchanged by the proposed amendments. The modified specifications which correct administrative errors and clarify existing Technical Specification requirements do not significantly reduce any of the margins of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Florida International University, University Park, Miami, FL 33199

Attorney for licensee: J. R. Newman, Esquire, Morgan, Lewis & Bockius, 1800 M Street, NW., Washington, DC 20036

NRC Project Director: David B. Matthews

Gulf States Utilities Company, Cajun Electric Power Cooperative, and Entergy Operations, Inc., Docket No. 50-458, River Bend Station, Unit 1, West Feliciana Parish, Louisiana

Date of amendment request: August 17, 1995

Description of amendment request: The proposed amendment would allow the containment to be opened after about 11 days following shutdown during refueling and would redefine the operability requirements for selected engineered safety feature systems such that these systems are only required to be operable during the calculated decay period. The proposed changes will not remove requirements for systems to mitigate potential vessel draindown events, will not remove requirements for systems required for decay heat removal, and will continue to require high water level over the vessel during fuel movement. Programs are in place to close the containment, if needed, to address shutdown risk concerns.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

The proposed limits on recently irradiated fuel is used to establish operational conditions where specific activities represent situations where significant radioactive releases can be postulated. These operational conditions are consistent with the design basis analysis. Because the equipment affected by the revised operational conditions is not considered an initiator to any previously analyzed accident, inoperability of the equipment cannot increase the probability of any previously evaluated accident.

The proposed applicability in conjunction with existing administrative controls on light loads, bounds the conditions of the current design basis fuel handling accident analysis. The analysis also concludes the limiting offsite radiological consequences are well within the acceptance criteria of NUREG 0800, Section 15.7.4 and GDC 19. The analysis is also conducted in a conservative manner containing margins in the calculation of mechanical analysis, iodine inventory and iodine decontamination factor. Each of these conservatisms will further decrease the consequences. Therefore, the proposed changes do not significantly increase the probability or consequences of any previously evaluated accident.

The proposed limits are used to establish operational conditions where specific activities represent situations where significant radioactive releases can be postulated. In addition, the changes to operation are consistent with previous limits -- only allowing increased flexibility after the radiological consequences are assured to remain within accepted limits. Therefore, these operational conditions are consistent with the design basis analysis. The proposed changes do not introduce any new modes of plant operation and do not involve physical modifications to the plant. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previous analyzed.

The revised limits are used to establish operational conditions where specific activities represent situations where significant radioactive release can be postulated. These operational conditions are consistent with the design basis analysis and are established such that the radiological consequences are at or below the current RBS licensing limit. Safety margins and analytical conservatisms have been evaluated and are well understood. Conservative methods of analysis are maintained through the use of accepted methodology and benchmarking the proposed methods to previous analysis. Margins are retained to ensure that the analysis adequately bounds all postulated event scenarios. The proposed change only eliminates some excess conservatism from the analysis.

EOI has implemented NUMARC 91-06 guidelines for shutdown operations at RBS. Shutdown Operations Protection Plan and Primary-Secondary Containment Integrity procedures presently include guidance for closure of the containment hatch and other significant opening in containment, in addition to the requirements contained in the license and design basis. This additional protection will enhance the ability to limit offsite effects.

Acceptance limits for the fuel handling accident are provided in 10CFR100 with additional guidance provided in NUREG 0800, Section 15.7.4 Excess margin is the difference between the postulated doses and the corresponding licensing limit. In the initial review of River Bend Station for operation (NUREG-0989, Section 15.7.4), the NRC accepted the design and analysis based on meeting the guideline dose limits of 10CFR100 and SRP 15.7.4. The proposed applicability continues to ensure that the whole-body and thyroid doses at the exclusion area and low population zone boundaries, as well as control room doses, are below the corresponding licensing limit. These margins are unchanged; therefore, the proposed changes do not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Government Documents Department, Louisiana State University, Baton Rouge, LA 70803

Attorney for licensee: Mark Wetterhahn, Esq., Winston & Strawn, 1400 L Street, N.W., Washington, DC 20005

NRC Project Director: William D. Beckner

Indiana Michigan Power Company, Docket Nos. 50-315 and 50-316, Donald C. Cook Nuclear Plant, Unit Nos. 1 and 2, Berrien County, Michigan

Date of amendment requests: June 20, 1995 (AEP:NRC:0692CX)

Description of amendment requests: The proposed amendments would remove the requirements for fire protection systems from the licenses and the Technical Specifications (T/S) in accordance with the provisions and guidance of Generic Letters (GL) 86-10, ''Implementation of Fire Protection Requirements,'' 88-12, ''Removal of Fire Protection Requirements from Technical Specifications,'' and 93-07, Modification of the Technical Specification Administrative Control Requirements for Emergency and Security Plans.''

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

We have evaluated the proposed T/S changes and have determined that the changes should involve no significant hazards consideration based on the criteria established in 10 CFR 50.92(c). Operation of CNP [Cook Nuclear Plant] in accordance with the proposed amendment will not satisfy any of the following criteria.

(a) Involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed changes are administrative in nature, in that it moves the T/Ss portion of the Fire Protection Program from the T/Ss to the UFSAR [Updated Final Safety Analysis Report] and the implementing procedures. This is accomplished by referencing in the UFSAR and the documents which address the Fire Protection Program in greater detail. Thus, the proposed changes will not revise the requirements for fire protection equipment operability, testing, or inspection, but only moves the T/Ss portion of the Fire Protection Program to implementing procedures.

As fire protection requirements are only being relocated following the guidance of GLs 86-10, 88-12, and 93-07, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

(b) Create the possibility of a new or different kind of accident from any previously analyzed.

The proposed changes do not involve any physical alteration of plant configurations, changes to setpoints, or operating parameters. [These] are administrative changes that retain the existing fire protection requirements and relocate these requirements from the T/S to the UFSAR; therefore, these changes do not create the possibility of a new or different kind of accident.

(c) Involve a significant reduction in a margin of safety.

The proposed changes follow guidance contained in GLs 86-10, 88-12, and 93-07 for incorporating the Fire Protection Program into the UFSAR. A license condition will be implemented that will require that no changes can be made to the Fire Protection Program that will adversely affect the ability to achieve or maintain safe shutdown in the event of a fire without prior NRC approval. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment requests involve no significant hazards consideration.

Local Public Document Room location: Maud Preston Palenske Memorial Library, 500 Market Street, St. Joseph, Michigan 49085

Attorney for licensee: Gerald Charnoff, Esq., Shaw, Pittman, Potts and Trowbridge, 2300 N Street, NW, Washington, DC 20037

NRC Project Director: John N. Hannon

Northeast Nuclear Energy Company, et al., Docket No. 50-336, Millstone Nuclear Power Station, Unit No. 2, New London, Connecticut

Date of amendment request: August 23, 1995

Description of amendment request: The proposed amendment would revise Technical Specifications Section 3.8.1.1 and the Bases for Section 3/4.8. The proposed amendment would extend the Allowed Outage Time (AOT) for an Emergency Diesel Generator (EDG) from 72 hours to 7 days.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration (SHC), which is presented below:

... NNECO concludes that these changes do not involve a significant hazards consideration since the proposed change satisfies the criteria of 10 CFR 50.92(c). That is, the proposed changes do not:

1. Involve a significant increase in the probability or consequences of an accident previously analyzed.

The EDGs supply backup power to the essential safety systems in the event of a Loss of Normal (offsite) Power. EDGs are not accident initiators. Therefore, this change does not involve an increase in the probability of any accident previously evaluated.

Although the EDGs provide backup power to components that help mitigate the consequences of accidents previously evaluated, the extension in the AOT does not affect any of the assumptions used in the deterministic evaluations of these accidents. Thus, this change will not increase the consequences of any accident previously analyzed.

The increase in the EDG AOT introduces the potential to increase the risk to the public since a longer time window provides an opportunity to perform additional preventive maintenance to the EDG while the plant is on-line. However, the extended AOT, by itself, does not necessarily increase risk. The increase in the risk depends on the total time during which an EDG was out of service and the other equipment that is concurrently out of service with the EDG. The total risk increase due to the EDG being out of service will not be significant since that risk increase is monitored and kept at acceptable levels in accordance with the risk monitor program.

Based on the above, the proposal to extend the AOT for the EDGs (Technical Specification 3.8.1) does not involve a significant increase in the probability or consequences of an accident previously analyzed.

2. Create the possibility of a new or different kind of accident from any previously analyzed.

The proposed change to extend the AOT for the EDGs (Technical Specification 3.8.1) does not alter the physical design, configuration, or method of operation of the plant. Therefore, the proposal does not create the possibility of a new or different kind of accident from any previously analyzed.

3. Involve a significant reduction in the margin of safety.

The proposed change to extend the AOT for the EDGs (Technical Specification 3.8.1) do not affect the Limiting Conditions for Operations or their bases. As a result, the deterministic analyses performed to establish the margin of safety are unaffected. Thus, the change does not involve a significant reduction in the margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Learning Resources Center, Three Rivers Community-Technical College, 574 New London Turnpike, Norwich, CT 06360.

Attorney for licensee: Lillian M. Cuoco, Esq., Senior Nuclear Counsel, Northeast Utilities Service Company, P.O. Box 270, Hartford, CT 06141-0270

NRC Project Director: Phillip F. McKee

Northeast Nuclear Energy Company, et al., Docket No. 50-336, Millstone Nuclear Power Station, Unit No. 2, New London, Connecticut

Date of amendment request: August 23, 1995

Description of amendment request: The proposed amendment would extend the Allowed Outage Time (AOT) for an inoperable Safety Injection Tank (SIT) from 1 hour to 24 hours, unless the SIT is inoperable due to either boron concentration not within its limits or an inoperable level or pressure instrument. For these two special cases, the proposed change extends the AOT for an inoperable SIT to 72 hours. In addition, the proposed amendment clarifies the completion times and conditions for action statements and the criteria for surveillance requirements.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration (SHC), which is presented below:

... NNECO concludes that these changes do not involve a significant hazards consideration since the proposed change satisfies the criteria in 10 CFR 50.92(c). That is, the proposed changes do not:

1. Involve a significant increase in the probability or consequences of an accident previously analyzed.

The Safety Injection Tanks (SITs) are passive components in the Emergency Core Cooling System that mitigate the consequences of a Loss of Coolant Accident (LOCA). As such, the SITs are not accident initiators. Therefore, this change does not involve an increase in the probability of any accident previously evaluated.

The increase in the AOT has the potential to increase the risk if it becomes necessary to stay on-line longer than one (1) hour with an inoperable SIT. However, the estimated risk impact is negligible.

The SITs inject borated water into the reactor vessel (via the cold legs) during the blowdown phase of a large break LOCA. The introduction of the inventory of borated water from all four (4) SITs is needed to ensure adequate reflooding of the core (i.e., minimize core damage) until the Engineered Safety Feature (ESF) pumps can provide adequate core cooling. The SITs also provide makeup water for the Reactor Coolant System (RCS) for smaller break LOCAs. The extension of the AOT does not affect any of the assumptions used in the deterministic evaluations of these accidents. Thus, this change will not increase the consequences of any accident previously evaluated.

The increased AOT extension to 72 hours, based solely on instrumentation (level and pressure) malfunction, also does not involve a significant increase in the consequences of an accident previously evaluated as endorsed by the NRC in NUREG-1366, ''Improvements to Technical Specifications Surveillance Requirements.''

The modification to the completion times and the modification of the Surveillance Requirements for volumetric changes in the SIT as a result of addition from the Refueling Water Storage Tank (RWST) also do not involve a significant increase in the consequences of any accident previously evaluated by the NRC in NUREG-1432, ''Standard Technical Specifications for Combustion Engineering Plants.''

Based on the above, the proposed changes to extend the AOT for an inoperable SIT, clarify action statements, and modify the criteria for surveillance requirements, do not involve a significant increase in the probability or consequences of an accident previously analyzed.

2. Create the possibility of a new or different kind of accident from any previously analyzed.

The proposed changes to extend the AOT for an inoperable SIT, clarify action statements, and modify the criteria for surveillance requirements, do not alter the physical design, configuration, or method of operation of the plant. Therefore, the proposal does not create the possibility of a new or different kind of accident from any previously analyzed.

3. Involve a significant reduction in the margin of safety.

The proposed changes to extend the AOT for an inoperable SIT, clarify action statements, and modify the criteria for surveillance requirements, do not affect the Limiting Conditions for Operations (LCOs) of the SITs or the bases of the LCOs. As a result, the deterministic analyses performed to establish the margin of safety are unaffected. Thus, the change does not involve a significant reduction in the margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Learning Resources Center, Three Rivers Community-Technical College, 574 New London Turnpike, Norwich, CT 06360.

Attorney for licensee: Lillian M. Cuoco, Esq., Senior Nuclear Counsel, Northeast Utilities Service Company, P.O. Box 270, Hartford, CT 06141-0270

NRC Project Director: Phillip F. McKee

Northern States Power Company, Docket Nos. 50-282 and 50-306, Prairie Island Nuclear Generating Plant, Unit Nos. 1 and 2, Goodhue County, Minnesota

Date of amendment requests: May 4, 1995.

Description of amendment requests: The proposed amendments would revise the pressurizer and main steam safety valve lift setting tolerances from plus or minus 1% to plus or minus 3%, revise the Safety Limit curves and revise the Technical Specification Section 2 to conform to Standard Technical Specifications.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. The proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated

The proposed changes increase the ''as-found'' setpoint tolerances for the Pressurizer Safety Valves and Main Steam Safety Valves from [plus or minus] 1% to [plus or minus] 3%. The proposed changes do not involve any hardware modifications to plant structures, systems, or components. Analyses have determined that the proposed changes do not significantly affect the structural integrity of either the Reactor Coolant System or the Main Steam system.

The proposed setpoint tolerance of [plus or minus] 3% was included in the assumptions for the performance of the reload safety evaluations for the current fuel cycles, PI1-17 and PI2-16, and subsequent Prairie Island fuel cycle analyses. These analyses concluded that the minimum acceptable DNBR [departure from nucleate boiling ratio] is maintained, over-pressure protection is maintained, LOCA [loss-of-coolant accident] acceptance criteria are met and offsite dose limits are not exceeded. These changes are consistent with the guidance provided by Section III and XI of the ASME [American Society of Mechanical Engineers] Code and Standard Technical Specifications.

The proposed change to Technical Specification Figure TS.2.1-1 does not affect any existing accident analyses. This revision ensures that the design bases and safety limits are accurately and appropriately reflected in the Technical Specifications and will ensure that plant operations are properly evaluated for DNBR encroachment.

Therefore, the probability or consequences of an accident previously evaluated are not affected by any of the proposed amendments.

2. The proposed amendment will not create the possibility of a new of different

kind of accident from any accident previously analyzed The lift setpoint the Pressurizer Safety Valves and Main Steam Safety Valves will be restored to [plus or minus] 1% following testing, thus the ''as-left'' setpoint tolerance for the Pressurizer Safety Valves and Main Steam Safety Valves are unchanged. Evaluations of plant normal operation, transient and accident conditions have been performed assuming these safety valve lift settings are [plus or minus] 3% of the nominal setpoint and demonstrated that new or different kinds of accidents are not created by the proposed changes.

The proposed changes to Technical Specification Figure TS.2.1-1 do not affect the design, function or operation of any Prairie Island structures, systems or components. The curves show the loci of points of reactor core differential temperature (an indication of thermal power), Reactor Coolant System pressure, and average temperature for which the minimum DNBR is not less than the safety analysis limit, that fuel centerline temperature remains below melting, that the average enthalpy in the hot leg is less than or equal to enthalpy of saturated liquid, or that the exit quality is within the limits defined by the applicable DNBR correlation. There are no new failure modes introduced by the proposed changes to the Figure. The changes conservatively adjust Figure TS.2.1-1 to current plant conditions and ensure that the design is accurately reflected and that the plant is operated in accordance with its design bases.

Therefore, the possibility of a new or different kind of accident from any accident previously evaluated would not be created [by] these amendments.

3. The proposed amendment will not involve a significant reduction in the margin of safety

The proposed changes to the safety valve lift setting tolerances are consistent with the guidance provided by Section III and XI of the ASME Code and Standard Technical Specifications. Analyses have demonstrated these valves will continue to perform their function of protecting their respective system from over-pressurization under all postulated transients and accidents. The changed setting tolerances do not cause a reduction in any other safety margin such as DNBR. SAFETY LIMIT curves are provided to define minimum allowable safety margin for plant steady state operation, normal operational transients and anticipated operational occurrences. The SAFETY LIMITs represent a design requirement for establishment of many of the RPS [reactor protection system] trip setpoints which prevent reactor conditions from approaching the SAFETY LIMITs. The proposed revision of the SAFETY LIMIT curves provide the minimum safety margins with somewhat more conservatism than previously included. No RPS trips setpoints are changed.

Therefore, a significant reduction in the margin of safety would not be involved with these amendments.

Based on the evaluation described above, and pursuant to 10 CFR Part 50, Section 50.91, Northern States Power Company has determined that operation [of] the Prairie Island Nuclear Generating Plant in accordance with the proposed license amendment request does not involve any significant hazards considerations as defined by Nuclear Regulatory Commission regulations in 10 CFR Part 50, Section 50.92.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment requests involve no significant hazards consideration.

Local Public Document Room location: Minneapolis Public Library, Technology and Science Department, 300 Nicollet Mall, Minneapolis, MN 55401

Attorney for licensee: Jay Silberg, Esq., Shaw, Pittman, Potts, and Trowbridge, 2300 N Street, NW, Washington, DC 20037

NRC Project Director: John N. Hannon

Pennsylvania Power and Light Company, Docket Nos. 50-387 and 50-388 Susquehanna Steam Electric Station, Units 1 and 2, Luzerne County, Pennsylvania

Date of amendment request: July 28, 1995

Description of amendment request: The proposed amendment would revise the 250 volt DC [direct current] profiles in Technical Specifications Surveillance 4.8.2.1 (d) (2c) to reflect the new load profile calculations.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated.

[Final Safety Analysis Report] FSAR Section 8.3 states that the station batteries have sufficient capacity without the charger to independently supply the required loads for four hours. The Technical Specifications require that the batteries be surveilled to dummy loads which are greater than the design loads. The load profiles for the 250 VDC batteries were recalculated using discrete increments of time when the loads would be in use for each of five design basis events. The Technical Specification load profiles are a composite of the worst case loads for the events plus margin. The required ampere-hours for each battery using the new load profiles is less than the ampere-hours required using the existing load profiles. Therefore, since the load profiles envelop the actual loads on the batteries, the change to the 250 VDC battery load profiles does not involve a significant increase in the probability or consequence of an accident previously evaluated.

2. Create the possibility of a new or different kind of accident from any accident previously evaluated.

As stated above, the 250 VDC batteries have sufficient capacity to power the actual battery loads thus enabling them to perform their intended function. Any postulated accident resulting from this change is bounded by previous analysis. Therefore, the change to the 250 VDC battery load profiles does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Involve a significant reduction in a margin of safety.

The Class 1E 250 VDC batteries are required to have sufficient capacity and capability to ensure sufficient power is available to supply the safety related equipment for (1) the safe shutdown of the facilities and (2) the mitigation and control of accident conditions within the facilities. The proposed load profiles envelope the worst case loads plus margin.

The ampere-hours removed from the Class 1E 250 VDC batteries are less for the proposed load profiles than the existing load profiles. The ampere-hours available in the batteries after the batteries have supplies[d] the emergency loads for 4 hours are: [See table in subject application].

* * * * * * *

Engineering calculation shows that the Class 1E 250 VDC batteries maintain at least 210 VDC at the Class 1E 250 VDC MCCs while supplying the proposed loads, corrected for temperature and aging. Since the Class 1E 250 VDC circuits are designed to operate properly with a minimum of 210 VDC at the Class 1E MCCs, all the Class 1E emergency equipment supplied from the Class 1E batteries have sufficient voltage to operate for 4 hours after the loss of ac power.

The Class 1E 250 VDC batteries and Class 1E 250 VDC battery chargers have been sized using the proposed load profiles. The Engineering calculation shows that the 120 cell, 12 positive plates per cell battery banks are sufficient to supply the proposed load profiles, corrected for temperature and aging. The same calculation also shows that the Class 1E 250 VDC battery chargers have sufficient capacity to re-charge the batteries from the proposed emergency discharged conditions to the fully charged condition in 12 hours while continuing to supply the plant normal continuous loads.

Base upon the above discussion, the proposed changes to the Technical Specification load profiles do not reduce the margin of safety as defined in the Technical Specifications.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Osterhout Free Library, Reference Department, 71 South Franklin Street, Wilkes-Barre, Pennsylvania 18701

Attorney for licensee: Jay Silberg, Esquire, Shaw, Pittman, Potts and

Trowbridge, 2300 N Street NW., Washington, DC 20037

NRC Project Director: John F. Stolz

Pennsylvania Power and Light Company, Docket No. 50-388, Susquehanna Steam Electric Station, Unit 2, Luzerne County, Pennsylvania

Date of amendment request: August 11, 1995

Description of amendment request: The proposed amendment would revise Susquehanna Unit 2 Technical Specification Table 3.3.7.5-1 as follows:a.

Revise Item ι13, Required Number of Channels from 1 to 2;b.

Revise Item ι13, Minimum Channel Operable from 0 to 1;c.

Delete Footnote ιι.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

I. This proposal does not involve a significant increase in the probability or consequences of an accident previously evaluated.

Reestablishing the channel operability values in Item ι13 of Technical Specification Table 3.3.7.5-1, and deleting footnote ιι, has no impact on the probability or consequences of an accident previously evaluated. The proposed change in the channel operability values is a return to the values which were reviewed as part of the licensing basis.

II. This proposal does not create the possibility of a new or different kind of accident from any accident previously evaluated.

Reestablishing the channel operability values in Item ι13 of Technical Specification Table 3.3.7.5-1, and deleting footnote ιι, does not create the possibility of a new or different kind of accident from any accident previously evaluated. The change in the channel operability values increases the required number of channels available for accident monitoring. There is no correlation between increasing the number of neutron flux accident monitoring channels available and the creation of accident scenarios.

III. This change does not involve a significant reduction in a margin of safety.

Reestablishing the channel operability values in Item ι13 of Technical Specification Table 3.3.7.5-1, and deleting footnote ιι, does not involve a reduction in a margin of safety. The proposed change increases the number of required channels from current levels, and restores the values to those which have historically been required. At the present time, the number of required channels is being administratively controlled at the proposed levels to ensure conservatism in operability.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Osterhout Free Library, Reference Department, 71 South Franklin Street, Wilkes-Barre, Pennsylvania 18701

Attorney for licensee: Jay Silberg, Esquire, Shaw, Pittman, Potts and Trowbridge, 2300 N Street NW., Washington, DC 20037

NRC Project Director: John F. Stolz

Power Authority of the State of New York, Docket No. 50-333, James A. FitzPatrick Nuclear Power Plant, Oswego County, New York

Date of amendment request: May 12, 1995

Description of amendment request: The proposed change would extend the surveillance test intervals for the emergency service water (ESW) system to support 24 month operating cycles.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

Operation of the FitzPatrick plant in accordance with the proposed Amendment would not involve a significant hazards consideration as defined in 10 CFR 50.92 since it would not:

1. involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed changes increase the interval between ESW system surveillance tests. These changes are consistent with the guidance provided in Generic Letter 91-04. These changes do not involve any physical changes to the plant, nor do they alter the typical way the ESW system functions. On-line testing will continue to assure equipment availability. The type of testing and the corrective actions required if the subject ESW surveillances fail remain the same. As such, the proposed changes create no new impacts on accidents previously evaluated.

Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed changes increase the interval between ESW system surveillance tests. These changes are consistent with the guidance provided in Generic Letter 91-04. The proposed changes do not change the ability of the ESW system to provide heat removal for the ECCS [emergency core cooling system] components and other equipment essential to reactor shutdown. Past equipment performance and on-line testing indicate the longer test intervals will not degrade ESW equipment. No changes are proposed to the type of testing performed, only to the length of the surveillance interval. The proposed changes do not modify the design or operation of plant equipment, therefore, no new or different failure modes are introduced.

Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. involve a significant reduction in a margin of safety.

The proposed changes increase the interval between ESW system surveillance tests. These changes are consistent with the guidance provided in Generic Letter 91-04. The proposed changes do not alter the configuration of the ESW system nor change the manner in which the ESW equipment functions. Past equipment performance and on-line testing indicate the longer test intervals will not degrade ESW equipment. Operation of the plant remains unchanged by the proposed changes.

Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Reference and Documents Department, Penfield Library, State University of New York, Oswego, NY 13126

Attorney for licensee: Mr. Charles M. Pratt, 1633 Broadway, New York, NY 10019

NRC Project Director: Ledyard B. Marsh

Power Authority of the State of New York, Docket No. 50-333, James A. FitzPatrick Nuclear Power Plant, Oswego County, New York

Date of amendment request: June 15, 1995

Description of amendment request: The proposed change would extend the surveillance test intervals for the control rod system to support 24 month operating cycles.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

Operation of the FitzPatrick plant in accordance with the proposed Amendment would not involve a significant hazards consideration as defined in 10 CFR 50.92, since it would not:

11. involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed changes increase the interval between control rod system surveillance tests. These changes are consistent with the guidance provided in Generic Letter 91-04. These changes do not involve any physical changes to the plant, nor do they alter the way the control rod system functions. The type of testing and the corrective actions required if the subject control rod surveillances fail remain the same. As such, the proposed changes create no new impacts on accidents previously evaluated.

The reactivity margin - core loading test can be safely extended to accommodate the 24 month operating cycle. The calculation of reactivity margin takes into account the longer operating cycle.

The control rod scram time test can be safely extended to accommodate a 24 month operating cycle. Operating experience has indicated that control rod scram times do not significantly change over an operating cycle. Additional on-line testing provides adequate assurance of equipment operability.

The SDIV [Scram Discharge Instrument Volume] vent and drain valve operability test can be safely extended to accommodate a 24 month operating cycle. Evaluation of past surveillance performance and additional on-line testing assure valve operability. The operability of the mode switch and the reset switch is demonstrated during shutdowns.

Therefore, the proposed changes do not involve a significant increase in the probability and do not change the consequences of an accident previously evaluated.

2. create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed changes increase the interval between control rod system surveillance tests. These changes are consistent with the guidance provided in Generic Letter 91-04. The proposed changes do not change the ability of the control rod system to provide rapid reactivity control in order that no fuel damage results from any abnormal operating transient. Past equipment performance and on-line testing indicate the longer test intervals will not degrade control equipment. No changes are proposed to the type of testing performed, only to the surveillance interval length. The proposed changes do not modify the design or operation of plant equipment, therefore, no new or different failure modes are introduced.

Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. involve a significant reduction in a margin of safety.

The proposed changes increase the interval between control rod system surveillance tests. These changes are consistent with the guidance provided in Generic Letter 91-04. The proposed changes do not alter the configuration of the control rod system nor change the manner in which the control rod system functions. Past equipment performance and on-line testing indicate the longer test intervals will not degrade control rod equipment. Operation of the plant remains unchanged by the proposed changes.

Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Reference and Documents Department, Penfield Library, State University of New York, Oswego, NY 13126

Attorney for licensee: Mr. Charles M. Pratt, 1633 Broadway, New York, NY 10019

NRC Project Director: Ledyard B. Marsh

Power Authority of the State of New York, Docket No. 50-333, James A. FitzPatrick Nuclear Power Plant, Oswego County, New York

Date of amendment request: July 21, 1995

Description of amendment request: The proposed changes would replace the title-specific list of members on the Plant Operating Review Committee (PORC) with a more general statement of membership requirements, similar to that used to define Safety Review Committee membership; expand the scope of disciplines represented on the PORC to include Nuclear Licensing and Quality Assurance; change several management position titles; and, make several editorial corrections to the Technical Specifications.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

Replacing the title specific list of PORC members with a statement of membership requirements for the committee does not reduce the effectiveness of the committee to advise the Resident Manager (Site Executive Officer) on matters regarding nuclear safety.

The proposed title changes for the Chief Nuclear Officer, Site Executive Officer, Shift Manager, and Control Room Supervisor are changes in title only and do not affect the responsibilities, authority, qualification requirements, or reporting relationships of these positions.

The change proposed for Specification 6.12 is administrative in nature, reflecting a change previously approved elsewhere in Technical Specifications.

The Radiological and Environmental Services Manager title change proposed for Specification 6.11(A)2 is administrative in nature, reflecting a change previously approved elsewhere in Technical Specifications.

The remainder of proposed changes correct grammar or improve consistency in Technical Specification formatting and do not affect the meaning or intent of the specifications involved.

Operation of the James A. FitzPatrick Nuclear Power Plant in accordance with the proposed amendment would not involve a significant hazards consideration as defined in 10 CFR 50.92. The changes are administrative in nature and would not:

1. involve a significant increase in the probability or consequences of an accident previously evaluated,

2. create the possibility of a new or different kind of accident from those previously evaluated, or

3. involve a significant reduction in the margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Reference and Documents Department, Penfield Library, State University of New York, Oswego, New York 13126.

Attorney for licensee: Mr. Charles M. Pratt, 1633 Broadway, New York, NY 10019

NRC Project Director: Ledyard B. Marsh

Public Service Electric & Gas Company, Docket No. 50-354, Hope Creek Generating Station, Salem County, New Jersey

Date of amendment request: July 27, 1995

Description of amendment request: The proposed change to the Technical Specifications (TS) would incorporate updated pressure vs. temperature operating limit curves contained in TS Figure 3.4.6.1-1 and revise TS Surveillance Requirement 4.4.6.1.3 based on implementation of Regulatory Guide 1.99, Rev. 2 in accordance with Generic Letter 88-11. The changes are a result of data obtained from the first set of specimen capsules removed during Refueling Outage 5.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. Will not involve a significant increase in the probability or consequences of an accident [...] previously evaluated.

The proposed changes assure that the existing safety limits are not exceeded due to changing Reactor Vessel conditions. These changes reflect the latest material testing results in accordance with 10CFR50, Appendix G. The proposed changes to the pressure and temperature limits do not increase the probability of nonductile failures. The proposed changes to the surveillance requirement and the associated changes to the Bases to include a commitment to the methodology of Regulatory Guide 1.99, Rev. 2 ensures that the most limiting Reactor Vessel material is used in the determination of the pressure-temperature operating limits.

Therefore, it may be concluded that the proposed changes do not involve a significant increase in the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated.

2. Will not create the possibility of a new or different kind of accident from any previously evaluated.

No physical plant modifications or new operating configurations result from these changes. These changes do not adversely affect the design or operation of any system or component important to safety, rather they establish limits to assure that operations remain within acceptable safety boundaries.

Therefore, the possibility of a new or different kind of accident from any previously evaluated will not be created.

3. Will not involve a significant reduction in a margin of safety. Analysis of the capsule specimens has concluded that the Reactor Vessel has sufficient fracture toughness for continued safe operation, provided that operation remains within acceptable pressure-temperature limits. The revised pressure-temperature curves define these acceptable pressure-temperature limits during plant operation. The proposed changes maintain the existing margins of safety by modifying the operating limits based on the most limiting of the actual reference temperature shifts. This new limit considered analytical results of the capsule specimens, or a predicted shift considering the most limiting pressure vessel material. Changes to the Surveillance Requirement criteria and the associated Bases to include a commitment to the methodology contained in Regulatory Guide 1.99, Rev. 2 will ensure that the most limiting plate or beltline weld material will be utilized in the determination of the pressure-temperature limits.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Pennsville Public Library, 190 S. Broadway, Pennsville, NJ 08070

Attorney for licensee: M. J. Wetterhahn, Esquire, Winston and Strawn, 1400 L Street, NW., Washington, DC 20005-3502

NRC Project Director: John F. Stolz

Southern California Edison Company, et al., Docket Nos. 50-361 and 50-362, San Onofre Nuclear Generating Station, Unit Nos. 2 and 3, San Diego County, California

Date of amendment requests: August 1, 1995

Description of amendment requests: The amendment request proposes to change Technical Specification (TS) 3/4.3.2, Table 3.3-3, ''Engineered Safety Features Actuation System Instrumentation.'' TS 3/4.3.2 includes the requirements for the minimum number of toxic gas isolation system (TGIS) trains operable. The TS change request is to extend the allowed TGIS outage times during replacement of TGIS instrumentation.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The Toxic Gas Isolation System (TGIS) is designed to monitor and mitigate the effects of toxic gas releases on control room habitability. TGIS unavailability is not a precursor to any accident previously evaluated in Chapter 15 of the San Onofre Updated Final Safety Analysis Report (UFSAR). A risk assessment of the TGIS instrumentation replacement activity was performed and found that the likelihood of a loss of control room habitability beyond that permitted by the Technical Specifications (TS) will not exceed 1E-6 over the duration of this TS change. In addition, a loss of control room habitability does not necessarily lead to an accident or core damage event. However, if a loss of control room habitability was conservatively assumed to lead to a core damage event, this increase in risk would still not constitute a significant increase in the consequences or probability of any accident previously evaluated since the increase is less than 3% of the average annual core damage risk from internal events as reported in the San Onofre Individual Plant Examination. Therefore, operation of the facility in accordance with this proposed change does not involve a significant increase in the probability or consequences of any accident previously evaluated.

2. The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

This change extends the allowed outage times of the TGIS system. The change does not affect the design or operation of any other plant systems. An increase in TGIS unavailability is not a precursor to any accident previously evaluated in Chapter 15 of the San Onofre UFSAR. Therefore, operation of the facility in accordance with this proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. The proposed change does not involve a significant reduction in a margin of safety.

During replacement of TGIS instrumentation a single channel of TGIS will be maintained operable except during periods when construction activity may result in spurious TGIS alarms. During these periods the control room will normally be isolated except for brief periods when the control room will be open to allow for air exchange or to allow for CREACUS equipment repair. These periods, when the control room is open without a TGIS channel available, will not exceed 54 hours during the entire period when this change is in effect. Operation with control room ventilation in the normal mode with a single channel of TGIS operable for 44 days and no TGIS channel available for up to 54 hours has been analyzed, and results in an increase in the probability of a loss of control room habitability which does not exceed 1E-6 over the duration of this TS change. Therefore, this proposed change does not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment requests involve no significant hazards consideration.

Local Public Document Room location: Main Library, University of California, P. O. Box 19557, Irvine, CA 92713

Attorney for licensee: T. E. Oubre, Esquire, Southern California Edison Company, P. O. Box 800, Rosemead, CA 91770

NRC Project Director: William H. Bateman

Notice Of Issuance Of Amendments To Facility Operating Licenses

During the period since publication of the last biweekly notice, the Commission has issued the following amendments. The Commission has determined for each of these amendments that the application complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment.

Notice of Consideration of Issuance of Amendment to Facility Operating License, Proposed No Significant Hazards Consideration Determination, and Opportunity for A Hearing in connection with these actions was published in the **Federal Register** as indicated.

Unless otherwise indicated, the Commission has determined that these amendments satisfy the criteria for categorical exclusion in accordance with 10 CFR 51.22. Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared for these amendments. If the Commission has prepared an environmental assessment under the special circumstances provision in 10 CFR 51.12(b) and has made a determination based on that assessment, it is so indicated.

For further details with respect to the action see (1) the applications for amendment, (2) the amendment, and (3) the Commission's related letter, Safety Evaluation and/or Environmental Assessment as indicated. All of these items are available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, and at the local public document rooms for the particular facilities involved.

Carolina Power & Light Company, et al., Docket No. 50-400, Shearon Harris Nuclear Power Plant, Unit 1, Wake and Chatham Counties, North Carolina

Date of application for amendment: October 24, 1994, as supplemented July 21, 1995. The July 21, 1995, letter provides clarification information and did not change the scope of the October 24, 1994, letter, or the initial no significant hazards consideration determination.

Brief description of amendment: The proposed amendment would revise the TS to allow the relocation of TS 3/4.3.7.12, Area Temperature Monitoring; and the associated Bases in the TS to licensee-controlled documents.

Date of issuance: August 28, 1995

Effective date: August 28, 1995

Amendment No.: 62

Facility Operating License No. NPF-63. Amendment revises the Technical Specifications.

Date of initial notice in **Federal Register:** November 23, 1994 (59 FR 60379) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 28, 1995.No significant hazards consideration comments received: No

Local Public Document Room location: Cameron Village Regional Library, 1930 Clark Avenue, Raleigh, NC 27605

Duke Power Company, et al., Docket Nos. 50-413 and 50-414, Catawba Nuclear Station, Units 1 and 2, York County, South Carolina

Date of application for amendments: May 18, 1995, as supplemented May 31, 1995

Brief description of amendments: The amendments revise the frequency for conducting the Catawba Unit 2 Integrated Leak Rate Test (ILRT) from a nominal frequency of once per 40 months to less than or equal to 70 months. This also involves the granting of an exemption from the requirements of 10 CFR Part 50, Appendix J, which is addressed by separate correspondence.

Date of issuance: August 18, 1995

Effective date: As of the date of issuance to be implemented within 30 days from the date of issuance.

Amendment Nos.: 133 and 127

Facility Operating License Nos. NPF-35 and NPF-52: Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** June 21, 1995 (60 FR 32362) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 18, 1995.No significant hazards consideration comments received: No

Local Public Document Room location: York County Library, 138 East Black Street, Rock Hill, SC 29730

Duquesne Light Company, et al., Docket No. 50-412, Beaver Valley Power Station, Unit 2, Shippingport, Pennsylvania

Date of application for amendment: April 26, 1995

Brief description of amendment: This amendment adds a requirement to Technical Specification (TS) 4.5.2.a to periodically verify that the High Head Safety Injection (HHSI) pump minimum flow valve, 2CHS*MOV373, is maintained open during plant operation in Modes 1, 2, and 3. Valve 2CHS*MOV373, must be maintained open to provide a minimum flowpath for the HHSI pumps thereby minimizing the likelihood of HHSI pump damage due to pump operation with insufficient flow. The amendment allows flexibility for local verification of valve position or flow indication if the control room indication is not available. Several editorial changes to TS 3/4.5.2 are also being made to provide consistent format with other TSs.

Date of issuance: August 25, 1995

Effective date: August 25, 1995

Amendment No.: 73

Facility Operating License No. NPF-73: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** June 6, 1995 (60 FR 29874). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 25, 1995. No significant hazards consideration comments received: No.

Local Public Document Room location: B. F. Jones Memorial Library, 663 Franklin Avenue, Aliquippa, PA 15001

Georgia Power Company, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, City of Dalton, Georgia, Docket No. 50-366, Edwin I. Hatch Nuclear Plant, Unit 2, Appling County, Georgia

Date of application for amendment: April 14, 1995, as supplemented by letters dated June 22 and July 18, 1995

Brief description of amendment: The amendment eliminates response time testing (RTT) requirements for selected sensors and specific loop instrumentations for (1) the Reactor Protection System (RPS), (2) the Isolation System, and (3) the Emergency Core Cooling System (ECCS). In addition, the Note for Surveillance Requirement 3.3.6.1.7, which reads: ‘‘Radiation detectors may be excluded,’’ is being removed since RTT is not required for any radiation detector that provides a primary containment isolation signal as indicated in Table 3.3.6.1-1 of the TS.

Date of issuance: August 23, 1995

Effective date: As of the date of issuance to be implemented within 60 days from the date of issuance

Amendment No.: 137

Facility Operating License No. NPF-5: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** July 5, 1995 (60 FR 35076) The June 22 and July 18, 1995, letters provided clarifying information that did not change the scope of the April 14, 1995, application and initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 23, 1995.No significant hazards consideration comments received: No

Local Public Document Room location: Appling County Public Library, 301 City Hall Drive, Baxley, GA 31513

Georgia Power Company, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, City of Dalton, Georgia, Docket Nos. 50-424 and 50-425, Vogtle Electric Generating Plant, Units 1 and 2, Burke County, Georgia

Date of application for amendments: January 3, 1995, as supplemented by letters dated June 14 and July 6, 1995.

Brief description of amendments: The amendments revise the Technical Specifications (TS) with editorial changes to the Action Statements of TS 3.8.1.1 and 3.8.1.2 in order to reflect the availability of a third offsite ac electrical source. Technical Specification 4.8.1.1.1 is clarified to specify that the offsite ac circuits connected to the onsite Class 1E distribution system are required to be verified OPERABLE. A footnote is added to TS 3.8.3.1 to allow the connection of the third offsite ac source to the onsite busses.

Date of issuance: August 29, 1995

Effective date: As of the date of issuance to be implemented within 30 days from the date of issuance

Amendment Nos.: 90 and 68

Facility Operating License Nos. NPF-68 and NPF-81: Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** February 1, 1995 (60 FR 6301) The June 14 and July 6, 1995, letters provided clarifying information that did not change the scope of the January 3, 1995, application and initial proposed no significant hazards consideration determination. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 29, 1995.No significant hazards consideration comments received: No

Local Public Document Room location: Burke County Library, 412 Fourth Street, Waynesboro, GA 30830

Florida Power and Light Company, Docket Nos. 50-250 and 50-251, Turkey Point Plant Units 3 and 4, Dade County, Florida

Date of application for amendments: May 23, 1995

Brief description of amendments: The amendments revise the column format for the Reactor Protection System and Engineered Safety Feature Actuation System Setpoints

Date of issuance: August 24, 1995

Effective date: August 24, 1995

Amendment Nos.: 176 and 170Facility Operating Licenses Nos. DPR-31 and DPR-41: Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** June 21, 1995 (60 FR 32364) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 24, 1995.No significant hazards consideration comments received: No

Local Public Document Room location: Florida International University, University Park, Miami, FL 33199

Indiana Michigan Power Company, Docket Nos. 50-315 and 50-316, Donald C. Cook Nuclear Plant, Unit Nos. 1 and 2, Berrien County, Michigan

Date of application for amendments: December 20, 1993, as supplemented July 19, 1994, and February 28, 1995.

Brief description of amendments: The amendments revise the surveillance requirements and load profiles for A, B, and N Train batteries.

Date of issuance: August 22, 1995

Effective date: August 22, 1995

Amendment Nos.: 198 and 183

Facility Operating License Nos. DPR-58 and DPR-74. Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** February 2, 1994 (59 FR 4939) and June 6, 1995 (60 FR 29879) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 22, 1995.No significant hazards consideration comments received: No.

Local Public Document Room location: Maud Preston Palenske Memorial Library, 500 Market Street, St. Joseph, MI 49085

Indiana Michigan Power Company, Docket Nos. 50-315 and 50-316, Donald C. Cook Nuclear Plant, Unit Nos. 1 and 2, Berrien County, Michigan

Date of application for amendments: May 25, 1995, and supplemented June 30, 1995

Brief description of amendments: The amendments allow fuel reconstitution when analyzed in accordance with NRC-approved methodologies. The amendments are line item improvements based on NRC Generic Letter 90-02, ''Alternative Requirements for Fuel Assemblies in Design Features Section of Technical Specifications,'' supplement 1.

Date of issuance: August 22, 1995

Effective date: August 22, 1995

Amendment Nos.: 199 and 184

Facility Operating License Nos. DPR-58 and DPR-74. Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** July 5, 1995 (60 FR 35081) The June 30, 1995, supplement provided a minor revision to the proposed Technical Specification pages which was within the scope of the original application and did not change the staff's initial proposed no significant hazards considerations determination. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 22, 1995.No significant hazards consideration comments received: No.

Local Public Document Room location: Maud Preston Palenske Memorial Library, 500 Market Street, St. Joseph, MI 49085

Maine Yankee Atomic Power Company, Docket No. 50-309, Maine Yankee Atomic Power Station, Lincoln County, Maine

Date of application for amendment: February 14, 1995

Brief description of amendment: This amendment makes the following administrative changes to the Maine Yankee (MY) Technical Specifications (TS):

a. Removes responsibility for audits of the emergency and security plans--including their implementing procedures--from the TS and assigns that responsibility to the emergency and security plans,

b. Assigns review responsibility for significant, accidental, unplanned, or uncontrolled radioactive releases to the Nuclear Safety Audit and Review (NSAR) Committee,

c. Assigns additional reporting requirements to the NSAR Committee, and

d. Provides the President of MY with the authority to initiate an audit of any area of facility operation.

Date of issuance: August 22, 1995

Effective date: As of the date of issuance, to be implemented within 30 days.

Amendment No.: 152

Facility Operating License No. DPR-36: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** March 29, 1995 (60 FR 16191) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 22, 1995.No significant hazards consideration comments received: No

Local Public Document Room location: Wiscasset Public Library, High Street, P.O. Box 367, Wiscasset, ME 04578

Northeast Nuclear Energy Company, Docket No. 50-245, Millstone Nuclear Power Station, Unit 1, New London County, Connecticut

Date of application for amendment: May 18, 1995

Brief description of amendment: The amendment revises the minimum temperature at which the reactor vessel head bolting studs are allowed to be placed under tension. In addition, the amendment revises the minimum reactor vessel metal temperature during core critical operation, revises the minimum reactor vessel metal temperature for pressure tests, makes editorial changes, and revises the Bases for the applicable section.

Date of issuance: August 23, 1995

Effective date: As of the date of issuance to be implemented immediately.

Amendment No.: 85

Facility Operating License No. DPR-21. Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** June 21, 1995 (60 FR 32369) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 23, 1995.No significant hazards consideration comments received: No.

Local Public Document Room location: Learning Resources Center, Three Rivers Community-Technical College, 574 New London Turnpike, Norwich, CT 06360

Northeast Nuclear Energy Company, Docket No. 50-245, Millstone Nuclear Power Station, Unit 1, New London County, Connecticut

Date of application for amendment: June 15, 1995

Brief description of amendment: The amendment changes the definition for an alteration of the reactor core to one that is consistent with the intent of the improved standard technical specifications. The amendment also makes administrative changes to several technical specification pages.

Date of issuance: August 28, 1995

Effective date: As of the date of issuance to be implemented within 30 days.

Amendment No.: 86

Facility Operating License No. DPR-21. Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** July 19, 1995 (60 FR 37097) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 28, 1995.No significant hazards consideration comments received: No.

Local Public Document Room location: Learning Resources Center, Three Rivers Community-Technical College, 574 New London Turnpike, Norwich, CT 06360

Northeast Nuclear Energy Company, et al., Docket No. 50-423, Millstone Nuclear Power Station, Unit No. 3, New London County, Connecticut

Date of application for amendment: April 28, 1995, as supplemented August 2, 1995.

Brief description of amendment: The amendment changes Technical Specification (TS) Sections 3.7.5, 4.7.5, and 3/4.7.5, to permit Millstone Unit 3 to remain in operation with the average ultimate heat sink water temperature greater than 75* F (but less than or equal to 77* F) for a period of 12 hours.

Date of issuance: August 28, 1995

Effective date: As of the date of issuance.

Amendment No.: 119

Facility Operating License No. NPF-49. Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** June 6, 1995 (60 FR 29881). The information in the licensee's submittal of August 2, 1995, did not require a change to the initial proposed no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 28, 1995.No significant hazards consideration comments received: No.

Local Public Document Room location: Learning Resources Center, Three Rivers Community-Technical College, 574 New London Turnpike, Norwich, CT 06360

Pacific Gas and Electric Company, Docket Nos. 50-275 and 50-323, Diablo Canyon Nuclear Power Plant, Unit Nos. 1 and 2, San Luis Obispo County, California

Date of application for amendments: June 29, 1995

Brief description of amendments: The amendments revise the combined Technical Specifications (TS) for Diablo Canyon Nuclear Power Plant, Unit Nos. 1 and 2 (DCPP) to add Mode 1 applicability to TS 3/4.4.2.2, ''Safety Valves - Operating,'' and changes the low-temperature overpressure protection (LTOP) system enable temperature for Mode 4 applicability from 323 degrees F to 270 degrees F in TS 3/4.3.2.1, ''Safety Valves - Shutdown.''

Date of issuance: August 23, 1995

Effective date: August 23, 1995

Amendment Nos.: Unit 1 - Amendment No. 107; Unit 2 - Amendment No. 106

Facility Operating License Nos. DPR-80 and DPR-82: The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** July 19, 1995 (60 FR 37098) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 23, 1995.No significant hazards consideration comments received: No.

Local Public Document Room location: California Polytechnic State University, Robert E. Kennedy Library, Government Documents and Maps Department, San Luis Obispo, CA 93407

Pennsylvania Power and Light Company, Docket Nos. 50-387 and 50-388 Susquehanna Steam Electric Station, Units 1 and 2, Luzerne County, Pennsylvania

Date of application for amendments: February 1, 1995, as supplemented by letter dated June 20, 1995

Brief description of amendments: The requested changes would modify the applicable operational conditions for the secondary containment isolation radiation monitors located on the refueling floor and for the monitor located in the railroad access shaft.

Date of issuance: August 24, 1995

Effective date: Both units, as of the date of issuance and is to be implemented within 30 days

Amendment Nos.: 152 and 122

Facility Operating License Nos. NPF-14 and NPF-22. The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** March 29, 1995 (60 FR 16192). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 24, 1995.No significant hazards consideration comments received: No

Local Public Document Room location: Osterhout Free Library, Reference Department, 71 South Franklin Street, Wilkes-Barre, PA 18701

Philadelphia Electric Company, Docket Nos. 50-352 and 50-353, Limerick Generating Station, Units 1 and 2, Montgomery County, Pennsylvania

Date of application for amendments: August 31, 1994, as supplemented by letters dated May 11, and July 3, 1995

Brief description of amendments: This amendment revises the Technical Specifications to permit the relocation of the Turbine Overspeed Protection System to the Updated Final Safety Analysis Report and Controlled Plant Procedures.

Date of issuance: August 24, 1995

Effective date: August 24, 1995

Amendment Nos.: 100 and 64

Facility Operating License Nos. NPF-39 and NPF-85. The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** November 9, 1994 (59 FR 55884) The supplemental letters do not change the initial no significant hazards consideration determination nor the initial Federal Register notice. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 24, 1995.No significant hazards consideration comments received: No

Local Public Document Room location: Pottstown Public Library, 500 High Street, Pottstown, PA 19464

Philadelphia Electric Company, Docket Nos. 50-352 and 50-353, Limerick Generating Station, Units 1 and 2, Montgomery County, Pennsylvania

Date of application for amendments: February 22, 1995

Brief description of amendments: The amendments revise the Technical Specifications Surveillance Requirements to clarify the Emergency Diesel Generator acceptable steady state voltage range.

Date of issuance: August 24, 1995

Effective date: As of the date of issuance and shall be implemented within 30 days of issuance.

Amendment Nos.: 101 and 65

Facility Operating License Nos. NPF-39 and NPF-85. The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** April 26, 1995 (60 FR 20525) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 24, 1995.No significant hazards consideration comments received: No

Local Public Document Room location: Pottstown Public Library, 500 High Street, Pottstown, PA 19464

Public Service Electric & Gas Company, Docket No. 50-354, Hope Creek Generating Station, Salem County, New Jersey

Date of application for amendment: January 13, 1995

Brief description of amendment: The amendment revised the Administrative Controls Section (6.0) of the Technical Specifications for Hope Creek Generating Station to reflect organizational changes and resultant management title changes.

Date of issuance: August 22, 1995

Effective date: August 22, 1995

Amendment No.: 77

Facility Operating License No. NPF-57: This amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** June 21, 1995 (60 FR 32371) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 22, 1995.No significant hazards consideration comments received: No

Local Public Document Room location: Pennsville Public Library, 190 S. Broadway, Pennsville, NJ 08070

Public Service Electric & Gas Company, Docket Nos. 50-272 and 50-311, Salem Nuclear Generating Station, Unit Nos. 1 and 2, Salem County, New Jersey

Date of application for amendments: December 23, 1994

Brief description of amendments: The amendments to the Technical Specifications revise the surveillance requirement to perform a visual inspection of containment areas affected by containment entry when containment integrity is established. They are consistent with Item 7.5 of Generic Letter 93-05, ''Line-Item Technical Specifications Improvements to Reduce Surveillance Requirements for Testing During Power Operation.''

Date of issuance: August 24, 1995

Effective date: As of the date of issuance, to be implementd within 60 days.

Amendment Nos.: 174 and 155

Facility Operating License Nos. DPR-70 and DPR-75. The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** February 1, 1995 (60 FR 6308) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 24, 1995.No significant hazards consideration comments received: No

Local Public Document Room location: Salem Free Public Library, 112 West Broadway, Salem, NJ 08079

Southern California Edison Company, et al., Docket Nos. 50-361 and 50-362, San Onofre Nuclear Generating Station, Unit Nos. 2 and 3, San Diego County, California

Date of application for amendments: September 16, 1994

Brief description of amendments: These amendments revise Technical Specification (TS) 3/4.2.1, ''Linear Heat Rate.'' The linear heat rate (LHR) limit for steady state operation is revised from 13.9 kw/ft to 13.0 kw/ft. The Bases for TS 3/4.2.1, ''Linear Heat Rate,'' is also being revised to reflect the new value.

Date of issuance: August 23, 1995

Effective date: August 23, 1995, to be implemented within 30 days of issuance.

Amendment Nos.: Unit 2 - Amendment No. 124; Unit 3 - Amendment No. 113

Facility Operating License Nos. NPF-10 and NPF-15: The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** November 9, 1994 (59 FR 55892) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 23, 1995. No significant hazards consideration comments received: No.

Local Public Document Room location: Main Library, University of California, P. O. Box 19557, Irvine, CA 92713

Tennessee Valley Authority, Docket Nos. 50-327 and 50-328, Sequoyah Nuclear Plant, Units 1 and 2, Hamilton County, Tennessee

Date of application for amendments: May 3, 1995

Brief description of amendments: The amendments delay implementation of Amendment Nos. 182 and 174 until implementation problems are addressed. These changes revise the setpoints and time delays for the auxiliary feedwater loss of power and the 6.9 kv shutdown board loss of voltage and degraded voltage instrumentation.

Date of issuance: August 22, 1995

Effective date: August 22, 1995

Amendment Nos.: 207 and 197

Facility Operating License Nos. DPR-77 and DPR-79: Amendments revise the technical specifications.

Date of initial notice in **Federal Register:** May 23, 1995 (60 FR 27343) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 22, 1995.No significant hazards consideration comments received: None

Local Public Document Room location: Chattanooga-Hamilton County Library, 1101 Broad Street, Chattanooga, TN 37402

Tennessee Valley Authority, Docket Nos. 50-327 and 50-328, Sequoyah Nuclear Plant, Units 1 and 2, Hamilton County, Tennessee

Date of application for amendments: April 6, 1995 (TS 94-18)

Brief description of amendments: The amendments revise Surveillance Requirement 4.0.5 by replacing the current Inservice Inspection program and the Inservice Testing program requirements with the requirements stated in the Standard Technical Specifications (NUREG-1431). The amendments also delete Technical Specification 3/4.4.10, ''Structural Integrity ASME Code Class 1, 2 and 3 Components,'' and its related Bases information.

Date of issuance: August 22, 1995

Effective date: August 22, 1995

Amendment Nos.: 208 and 198

Facility Operating License Nos. DPR-77 and DPR-79: Amendments revise the technical specifications.

Date of initial notice in **Federal Register:** April 26, 1995 (60 FR 20528)

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 22, 1995.No significant hazards consideration comments received: None

Local Public Document Room location: Chattanooga-Hamilton County Library, 1101 Broad Street, Chattanooga, TN 37402

Union Electric Company, Docket No. 50-483, Callaway Plant, Unit 1, Callaway County, Missouri

Date of application for amendment: April 17, 1995, as supplemented on June 30, 1995

Brief description of amendment: The amendment revises Technical Specifications Technical Specification 2.2.1, Table 2.2-1. The changes address reducing repeated alarms and partial reactor trips by revising the Overpower Delta-T setpoint function.

Date of issuance: August 21, 1995

Effective date: Immediately, to be implemented within 30 days.

Amendment No.: 102

Facility Operating License No. NPF-30. Amendment revises the Technical Specifications.

Date of initial notice in **Federal Register:** May 10, 1995 (60 FR 24922). The June 30, 1995, letter provided supplemental information that did not change the initial proposed no significant hazards consideration determination.The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 21, 1995. No significant hazards consideration comments received: No.

Local Public Document Room location: Callaway County Public Library, 710 Court Street, Fulton, MO 65251

Washington Public Power Supply System, Docket No. 50-397, Nuclear Project No. 2, Benton County, Washington

Date of application for amendment: September 2, 1992

Brief description of amendment: The amendment revises the required signal-to-noise ratio for the source range monitors, as recommended by General Electric.

Date of issuance: August 23, 1995

Effective date: August 23, 1995

Amendment No.: 140

Facility Operating License No. NPF-21: The amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** July 19, 1995 (60 FR 37101) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 23, 1995.No significant hazards consideration comments received: No.

Local Public Document Room location: Richland Public Library, 955 Northgate Street, Richland, WA 99352

Dated at Rockville, Maryland, this 6th day of September 1995.

For the Nuclear Regulatory Commission

Jack W. Roe,

Director, Division of Reactor Projects - III/IV Office of Nuclear Reactor Regulation.

[Doc. 95-22616 Filed 9-12-95; 8:45 am]

BILLING CODE 7590-01-F

OFFICE OF MANAGEMENT AND BUDGET

Managerial Cost Accounting Concepts and Standards

AGENCY: Office of Management and Budget.

ACTION: Notice of document availability.

SUMMARY: This Notice indicates the availability of the fourth Statement of Federal Financial Accounting Standards, ''Managerial Cost Accounting Concepts and Standards for the Federal Government,'' adopted by the Office of Management and Budget (OMB). The statement was recommended by the Federal Accounting Standards Advisory Board and adopted in its entirety by OMB.

ADDRESSES: Copies of the Statement of Federal Financial Accounting Standards No. 4, ''Managerial Cost Accounting Concepts and Standards for the Federal Government,'' may be obtained for $7.50 each from the Superintendent of Documents, Government Printing Office, Washington, DC 20402–9325 (telephone 202–783–3238), Stock No. 041–001–00457–2.

FOR FURTHER INFORMATION CONTACT: Ronald Longo (telephone: 202–395–3993), Office of Federal Financial Management, Office of Management and Budget, 725–17th Street, N.W.—Room 6025, Washington, DC 20503.

SUPPLEMENTARY INFORMATION: This Notice indicates the availability of the fourth Statement of Federal Financial Accounting Standards, ''Managerial Cost Accounting Concepts and Standards for the Federal Government.'' The standard was recommended by the Federal Accounting Standards Advisory Board (FASAB) in June 1995, and adopted in its entirety by the Office of Management and Budget (OMB).

Under a Memorandum of Understanding among the General Accounting Office, the Department of the Treasury, and OMB on Federal Government Accounting Standards, the Comptroller General, the Secretary of the Treasury, and the Director of OMB decide upon principles and standards after considering the recommendations of FASAB. After agreement to specific principles and standards, they are to be published in the **Federal Register** and distributed throughout the Federal Government.

G. Edward DeSeve,

Controller.

[FR Doc. 95–22766 Filed 9–12–95; 8:45 am]

BILLING CODE 3110–01–P

PENSION BENEFIT GUARANTY CORPORATION

Request for Extension of Approval Under the Paperwork Reduction Act; Collection of Information Under 29 CFR Part 2646, Reduction or Waiver of Partial Withdrawal Liability

AGENCY: Pension Benefit Guaranty Corporation.

ACTION: Notice of request for extension of OMB approval.

SUMMARY: The Pension Benefit Guaranty Corporation has requested that the Office of Management and Budget extend approval, under the Paperwork Reduction Act, of the collection of information requirements (1212–0039) contained in its regulation on Reduction or Waiver of Partial Withdrawal Liability (29 CFR Part 2646). The effect of this notice is to advise the public of the PBGC's request.

DATES: The PBGC is requesting that OMB complete action on the PBGC's request by September 29, 1995. Comments must be received by September 25, 1995.

ADDRESSES: All written comments should be addressed to: Office of Information and Regulatory Affairs, Office of Management and Budget, Attention: Desk Officer for Pension Benefit Guaranty Corporation, Washington, DC 20503. The request for extension will be available for public inspection at the PBGC's Communications and Public Affairs Department, Suite 240, 1200 K Street, NW., Washington, DC 20005–4026, between 9:00 a.m. and 4:00 p.m. on business days.

FOR FURTHER INFORMATION CONTACT: Deborah C. Murphy, Attorney, Office of General Counsel, Pension Benefit Guaranty Corporation, 1200 K Street, NW., Washington, DC 20005–4026, 202–326–4024 (202–326–4179 for TTY and TDD).

SUPPLEMENTARY INFORMATION: This collection of information is contained in the Pension Benefit Guaranty Corporation's regulation on Reduction or Waiver of Partial Withdrawal